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SUBJECT: Forwards Vols 1 & 2 of "Procedure Generation Package," per guidelines of NUREG-0737,Suppl 1.Evaluations of emergency procedures re ATWS & deviations from BWR Owners Group emergency program guidelines encl.

see '85 Reports

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MAY 13 1985

Director of Nuclear Reactor Regulation
Attention: Mr. W. R. Butler, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
EMERGENCY PROCEDURES GENERATION PACKAGE
ER 100450 FILE 842-03 Docket Nos. 50-387
PLA-2468 50-388

Reference: NRC letter, A. Schwencer to N. W. Curtis, "Order Confirming Licensee Commitments on Emergency Response Capability," dated June 14, 1984.

Dear Mr. Butler:

In accordance with item 4a of the attachment to the reference for Unit 1 and with item (d)(1) of Attachment 2 to the Unit 2 Operating License (NPF-22), enclosed please find the SSES Procedures Generation Package (PGP). The PGP is consistent with the guidance in NUREG 0737 Supplement 1. The PGP provides the basis for the development of plant-specific Emergency Operating Procedures which we intend to implement on July 12.

Also attached are two reports, "Evaluation of Susquehanna Steam Electric Station Emergency Procedures Relating to ATWS" and "Evaluation of SSES Deviations from BWR Owners Group EPG's (Non-ATWS)." These reports address deviations from the BWROG EPG's.

We trust this information is deemed acceptable. Should you require additional information, please contact W. E. Barberich (215-770-7850).

Very truly yours,

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

Attachment

cc: M. J. Campagnone NRC
R. H. Jacobs NRC

A003
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Evaluation of Susquehanna Steam Electric Station Emergency Procedures Relating to ATWS

1.0 Summary

The BWROG EPG's have been evaluated to determine their suitability for guidance in formulation of SSES response procedures for ATWS. The evaluations performed have identified a need for modifications to the BWROG guidelines as applied to ATWS response. These modifications involve the guidance for control of makeup water flow and vessel water level. Transient calculations have been performed which demonstrate that peak suppression pool temperatures remain acceptable when HPCI/RCIC flow is not terminated after L2 initiation. Under these conditions the water level will continue to fall below L2 until an equilibrium level is reached. This equilibrium level depends on the initiation time and rate of boron injection ranging between -60 inches to -95 inches relative to instrument zero. The level may be stabilized at this equilibrium value as boron is injected to prevent an increase in water level and the higher values of reactor power which result. The calculations also show that if the operator fails to accomplish longer term maintenance of this initial minimum level the consequences to suppression pool temperature remain acceptable. This is an important result since the transient calculations show that termination of flow, restart of flow and maintenance of level all would be required actions in the critical early period of the transient. The elimination of the need for these actions is believed to increase the overall probability of proper and timely operator actions. A further benefit is the finding that operation at higher water levels which avoid the downcomer area reduction from 300 ft^2 to 88 ft^2 at -110 inches is acceptable. This area reduction in combination with the trend toward increasing power to flow ratios as level decreases raises concern over the probability and severity of limit cycle operation which could have adverse consequences on ATWS mitigation capability. Maintaining level above -95 inches, or more likely -65 inches, greatly alleviates this concern. Finally, maintaining level well above -150 inches avoids loss of wide range indication and avoids level 1 initiation signals which could interfere with ATWS mitigating actions.

The procedures which will be adopted for SSES must initially apply to the current 43 gpm boron injection capability of the SLCS. Our calculations show that we have little or no margin with the existing HCTL curve regardless of response strategy with the 43 gpm injection rate. For this reason we also propose a new HCTL curve (to be used for ATWS only) which takes credit for the improved capability of the SRV discharge quenchers to condense steam without excessive condensation loads at elevated pool temperatures.

With this new HCTL curve we predict acceptable ATWS response for all evaluated level response strategies including those strategies which avoid loss of wide range level indication, avoid initiation of level 1 signals, and which minimize the likelihood of limit cycle operation.

2.0 Introduction

The objective of this document is to evaluate the application of the generic BWROG Emergency Procedure Guidelines to SSES specific ATWS sequences to assure that application of those guidelines will result in acceptable consequences to the plant even for the most limiting ATWS cases. The modifications evaluated herein comply with the intent of the BWROG EPGs. These modifications take

into account, however, the specific characteristics of SSES and consequently represent a deviation from a literal interpretation of the BWROG EPGs. These deviations are specific to ATWS events and are not applicable to any other accident sequence.

In the description of ATWS event sequences the operator actions discussed in this document do not represent all operator actions. The presentation will focus only on those where we believe special guidance is needed and which may not be readily deduced from the information provided by the BWROG EPG's.

Finally, the response of the plant as it is currently configured will be considered, as well as for the plant configuration which will exist after ATWS related equipment modifications are made. The primary difference, relative to transient response of the plant, is the current 43 gpm boron solution injection rate as opposed to the future 86 gpm injection rate.

The analysis required to support the ATWS response procedures presented here include:

1. Rationale and calculations to increase the Heat Capacity Temperature Limit of the suppression pool to 208^{o}F at operating pressure.

2. The water level control strategy best suited for response to a limiting ATWS event.

We believe that the procedural modifications that are discussed herein reduce the probability of conditions which could exceed containment integrity limits or require depressurization of a critical reactor to a small fraction of severe ATWS events. Further, severe ATWS events represent only a small fraction of all failure to scram events.

3.0 Description of Severe ATWS Sequences

The complete spectrum of potential ATWS event sequences is very broad. The intent here will be to identify the limiting ATWS sequences and to demonstrate that these limiting sequences can be successfully terminated without damage to the plant, utilizing the plant equipment and procedures recommended in this document.

Two types of severe ATWS events are considered, the isolation case and the non-isolation case. Of these two only the isolation case will be analyzed on the basis that it is more limiting than the non-isolation case. A description of the event sequences and operator actions are presented for both cases, however.

For all cases considered it is postulated that the most severe ATWS sequence is one which occurs at a time when the reactor is operating at full power and no rods insert when the turbine is lost due to stop valve or MSIV closure. Other than determining whether or not the event results in isolation, the nature of the transient initiator plays a minor role in determining the severity of the ATWS event. In this regard, only the suppression pool temperature is considered as a criterion for severity. The short term pressure and power transient occurring in the first few seconds of an ATWS event are considered acceptable for all event sequences.

The important issues relating to adequate mitigation of an ATWS sequence are:

1. Maintaining adequate liquid inventory in the vessel to assure core cooling.

2. Avoidance of severe power transients which could result in fuel clad perforations.

3. Power reduction to stabilize suppression pool temperature below a value which could lead to loss of containment integrity or loss of equipment vital to bringing the plant to a safe and stable condition or which could require depressurization of the reactor before hot shutdown.

The level of success in ATWS mitigation is determined by the performance of plant equipment and of the operators. If equipment and operator performance is consistent with equipment design, response procedures, and training, likelihood of success in mitigating ATWS events is quite high.

3.1 The Isolation ATWS

In the event of a transient without scram at full reactor power which requires reactor shutdown to avoid plant damage, the plant will undergo a transient which will result in operation at a new steady state condition after a short period of time even though no control rods insert. For example, in the case of a transient involving MSIV closure as the event requiring reactor trip, failure to trip will cause a rapid pressure and power increase in the first several seconds of the transient. This pressure transient will cause more than two relief valves to lift which will assure a pressure sufficient to trip the reactor recirculation pumps. These pumps will then coast down and the core flow will decrease to the natural circulation flow rate.

As the MSIVs close, the steam flow to the feedwater drive turbines will be lost and the recirculation pump coast down will be accompanied by a feedwater flow coastdown with eventual complete loss of feedwater flow. Since the reactor power will be producing steam in excess of the makeup rate of water to the vessel, the reactor water level will fall and the natural circulation flow rate will decrease. The reactor power is reduced as core flow rate is reduced so that the rate of loss of vessel water inventory decreases as level falls.

When level 2 is reached the HPCI and RCIC systems are initiated and these systems will supply 5600 gpm to the vessel. The level will continue to fall beyond level 2 until the reactor power generates steam at a rate which just matches this makeup flow rate. Calculations show that this condition will be achieved at a water level about 4 feet below level 2 when boron injection has not been initiated.

At this time the reactor will settle into a steady state operating condition with three relief valves continuously open and a fourth valve opening intermittently. Operation in this state can only be tolerated for a short period of time before the ATWS related suppression pool temperature limit is reached. Operator action to achieve reactor shutdown is essential if ATWS mitigation success criteria are to be satisfied. The operator can attempt to either manually scram the reactor or take action to manually insert the control rods individually.

In the case of a failure to scram from full power with isolation, there is essentially no time for individual rod insertion, and only initiation of SLCS can assure compliance with ATWS mitigation success criteria when automatic and manual reactor trips have failed. For one pump injection by the SLCS at 43 gpm about 1700 seconds will be required to inject sufficient boron to assure hot shutdown of the reactor. Two pump injection at 86 gpm would require 850 seconds to inject hot shutdown boron.

In the case of 43 gpm injection, SLCS initiation must occur within the first few minutes of the transient to be certain to meet the suppression pool temperature limit criterion. A significant delay can be accepted for 86 gpm injection. When hot shutdown conditions are achieved, however, heating of the suppression pool is not terminated due to decay heat. The decay heat level is roughly 1.5% of design power at the time of hot shutdown. An increase of pool temperature above the ATWS limit can be avoided if both loops of RHR are placed in the suppression pool cooling mode at the time of hot shutdown. In some cases a single loop of RHR will be sufficient to avoid the ATWS pool temperature limit.

For the isolation case, the essential operator action is immediate initiation of SLCS.

3.2 The Non-Isolation ATWS

The non-isolation ATWS involves a turbine trip without isolation. The turbine bypass valves are activated immediately as the turbine control or stop valves close, but they only have 25% of full steam flow capacity so that the reactor pressure rises as in the isolation case and relief valve operation is required. Since the two low set valves cannot handle the necessary steam flow, the reactor pressure is certain to rise to the set point of the second pressure group which assures trip of the recirculation pumps as in the isolation case.

For non-isolation, however, the feedwater system remains operational, and the level control system attempts to maintain normal water level (NWL). The feedwater system has the capacity to supply as much flow as is needed to assure maintaining NWL. The flow rate needed to accomplish this considerably exceeds the combined flow capacity of the HPCI and RCIC, so that when the system comes to natural circulation steady state conditions, the steaming rate is greater than in the case of the isolation event.

If the feedwater enthalpy were to remain constant, the steady state reactor power and steaming rate would be about 55% of the design values at a core flow rate of about 35 to 40 percent of design flow. Actually, since the turbine has tripped, extraction flow to the feedwater heaters is terminated, and so feedwater enthalpy will decrease at a rate depending on the thermal capacity of the feedwater trains. This in turn will result in a steady decrease of the steaming rate to the suppression pool, but not necessarily to a decrease of reactor power.

Nevertheless, the initial steaming rate may far exceed the bypass system capacity and suppression pool heating rates could exceed those for the isolation case by a considerable margin.

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If HPCI plus RCIC flow are adequate to allow successful termination of an ATWS event without damage to the plant for the isolation case, the obvious response would be to reduce feedwater flow to a value corresponding to HPCI plus RCIC flow. This action, if promptly taken, would reduce the non-isolation event to an event allowing an extended period of time for further operator action. This occurs since HPCI plus RCIC flow can only support a 21% steaming rate and this is within the 25% flow capacity of the bypass system. Therefore, immediate runback of feedwater flow will terminate suppression pool heating and allow an extended period for further operator actions.

In the event that the bypass system is unavailable, the event can readily be converted to the equivalent of the isolation ATWS simply by the runback of feedwater flow.

For the non-isolation case, the essential operator action is the runback of feedwater flow to the equivalent of HPCI plus RCIC flow. If bypass is not available he must also immediately start SLCS.

4.0 BWR Owner's Group Guidance for Operator Response to ATWS

The BWROG guidance for operator response to ATWS for isolation or non-isolation events involves the following actions.

1. Initiate SLCS when the suppression pool temperature reaches 110^{o}F.

2. Terminate all flow into the vessel except for SLCS and CRD cooling flow.

3. When level has fallen to TAF use high pressure injection systems to maintain level at TAF.

4. When a mass of boron solution sufficient for hot shutdown (HSD) has been injected, increase water level to promote boron mixing and achieve reactor shutdown.

5. Initiate suppression pool cooling (SPC).

5.0 Modification for SSES ATWS Response Procedures

The various BWROG EPG guidance items identified in section 4.0 are discussed individually below.

5.1 SLCS Initiation

The operator should be able to reliably identify an ATWS event within 2 minutes. For 43 gpm injection SLCS initiation should be immediate with no delay for any reason in isolation events. For the 86 gpm injection rate, some delay beyond 2 minutes could be allowed for SLCS initiation. The additional time allowed by the 86 gpm injection rate provides margin to allow for degraded equipment or operator performance.

5.2 Termination of Makeup Flow

Termination of makeup flow by HPCI/RCIC for the isolation case is unnecessary. The level stabilizes without operator action and the resulting steaming rate to the suppression pool does not lead to unacceptable pool temperatures. Elimination of the need to terminate HPCI/RCIC flow avoids concern over the potential for failure to achieve timely restart and frees the operator from the need for additional actions (trip and restart) in a critical time period.

For non-isolation cases the equivalent action would be trip of two out of three feedwater pumps and flow adjustment on the third to HPCI/RCIC equivalent flow. In this case the feedwater runback replaces initiation of SLCS as the critical action. Successful feedwater runback should essentially terminate suppression pool heating.

A related concern is the instruction to terminate HPCI/RCIC injection if RPV water level indication becomes unavailable. This is also unnecessary. The intent of the EPG guidance is to prevent the introduction of water to the turbines, but this is considered loss of an overall risk than rapidly adding a large quantity of heat to the suppression pool. That would be necessary to reduce RPV pressure below the shutoff head of low pressure pumps.

5.3 Maintain Level at TAF

For all isolation cases we recommend that HPCI and RCIC be allowed to initiate at L2 and continue to operate at full flow until the water level has fallen to its lowest level and stabilized or started to rise as boron injection reduces power. The minimum level for SSES should be about two to three feet below L2. The stable level in the absence of boron injection is believed to be about four feet below L2.

When the level has reached minimum the operator may choose to hold level constant at that value by throttling HPCI flow as needed or may allow full flow with a consequent increase in water level. Calculations have shown that the choice is not critical to successful ATWS mitigation.

Makeup flow should not be reduced to allow operation below the levels identified above. The reasons for this are:

1. At about five feet above TAF the downcomer free area reduces from 300 ft^2 to 88 ft^2. This aggravates concerns over limit cycle operation.

2. At just below L1 the wide range indication goes down scale and indication is lost. Only the fuel range indication is left, and it is not calibrated for Modes 1, 2, or 3.

3. At some level above L1 as specified in the technical specifications the L1 isolation signals will be generated. This could occur as high as two to three feet above TAF.

The level control strategy outlined above provides several major benefits. These are:

1. It reduces requirements for operator action in a critical time period.

2. It minimizes the chances of an apparent loss of level indication with the resultant need to depressurize.

3. It avoids taking the reactor further into a region where flow is reducing more rapidly than power, reducing the chances of unacceptable limit cycle operation. Such a trend would be further aggravated by a large factor by the reduction in downcomer area.

Calculations show that the resulting peak pool temperatures are not less acceptable than those that result from operation with level at TAF.

For non-isolation cases it is important to run back feedwater flow as quickly as possible. Even though bypass is available, the short term equilibrium steaming rate will exceed its capacity and could result in suppression pool heating rates as much as 50% greater than for the isolation case. If two feedwater pumps are tripped and the third is carefully run back to the equivalent of HPCI plus RCIC, the event becomes relatively benign in that the bypass can fully accommodate the steaming rate. This would reduce the pool heating rate to that associated with HPCI and RCIC turbine operation. These two systems should be allowed to operate at minimum flow with appropriate adjustments to the feedwater flow rate. This strategy will result in a level response very similar to that of the isolation cases. The benefits for the isolation case also apply to the non-isolation case.

5.4 Boron Remixing

There is no objection to this guidance in that calculations indicate that the result seems always favorable. In those cases where HPCI flow is not throttled, the remixing step need not be taken in that core flow is always maximized by maintaining downcomer level at the highest value within the HPCI capability.

5.5 Initiate Suppression Pool Cooling

There is no objection to this guidance. It is clear that the sooner SPC is initiated, the lower will be the peak pool temperature. Guidance should instruct that both loops of RHR be used for this purpose in order to minimize the rise in pool temperature after HSD has been achieved. A reasonable time interval to accomplish SPC initiation should be within eight to ten minutes after the start of the transient for the isolation case. For non-isolation cases with bypass, SPC initiation timing is not critical. It should be initiated in time to prevent significant pool temperature rise from HPCI and RCIC turbine exhaust however.

A high drywell pressure signal or a L1 signal will terminate SPC by opening the heat exchanger bypass valve and aligning the system to operate in the LPCI mode. With the reactor at pressure no injection will occur. Flow through the heat exchanger will be reduced from 10,000 gpm to 5000 gpm until the bypass valve is closed. A timer prevents reclosing the bypass valve for 10 minutes. For this reason SPC effectiveness will be reduced for a 10 minute period. While L1 is not expected to occur, calculations show that high drywell

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pressure is expected to occur at a suppression pool temperature in the range of 135°F to 145°F. Operator action is required to restore and maintain SPC at full capacity.

6.0 The Objectives of Operator Actions in Response to Severe ATWS Occurrences

Even with ideal response of the operators and equipment to a severe ATWS event, the plant will operate under conditions which have never been tested or thoroughly analyzed. The major concern relating to plant response to a severe ATWS event is the fact that natural circulation operation at low water level and with cold makeup water cannot be avoided. Calculations indicate that under these conditions core flow decreases more rapidly than core power as water level is decreased. This means that the ratio of core power to flow increases as water level falls. This ratio is an indication of the vulnerability of the reactor to entering a limit cycle mode of operation. Avoidance of limit cycle operation should be an objective of the mitigating actions taken in response to an ATWS event. Operation under limit cycle conditions could result in damage to the reactor, damage to the fuel or effective loss of instrumentation.

Calculations of plant response to an ATWS event have been made on the basis that limit cycle operation will be avoided. There is no assurance that this is true, however, and the further water level is decreased, the greater the risk of encountering limit cycle operation. The risk is believed to be considerably increased as water level is lowered below the top of the upper plenum due to the decrease in the downcomer free area by a factor of 3.4.

The primary objective of the operator must be to reach HSD before the suppression pool temperature limit (HCTL) is reached, and, if this is done, his response must be judged successful. An attempt to increase the margin by which the limiting temperature can be avoided by taking the reactor further into an untested range of operation should not be considered. Calculations have shown that there is only limited benefit in reducing water level below the minimum equilibrium level that can be supported by HPCI plus RCIC in any event.

7.0 Computational Models

The computational models used to determine an ATWS curve for HCTL and to follow ATWS shutdown transients are discussed here.

7.1 The Heat Capacity Temperature Limit

The origin of the requirement for reactor depressurization above some suppression pool temperature is the concern over severe dynamic loads due to SRV discharge phenomena. The Heat Capacity Temperature Limit (HCTL) curve is based on an upper limit the pool temperatures at which tests have been carried out and this temperature has been set as the maximum allowable pool temperature for any time when SRV discharge can occur. The limit is based on lack of positive knowledge that higher temperatures would result in acceptable dynamic loads rather than positive evidence to the contrary. Test data does exist which supports acceptable dynamic loads of pool temperatures approaching saturation. This is a credible result in that quencher submergence will always produce some degree of local subcooling and on physical grounds

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subcooling should be a critical parameter to determine the potential for severe dynamic loads.

We propose a new criterion be established for acceptable SRV dynamic loads under ATWS conditions. The recommendation is that all conditions under which 10^{o}F subcooling exists in the general pool volume at the quencher elevation will be considered to result in acceptable dynamic loads. There are two considerations in support of this change. First, it is reasonable to believe that a smooth steam quenching process is primarily dependent on local subcooling, and second, the risk associated with the chance of severe dynamic loads using this criterion under ATWS conditions is judged to be more acceptable than that risk associated with depressurizing a critical reactor with a resulting uncovering of fuel rods and subsequent entry of cold makeup water into the core.

For this purpose we conservatively assume a normal pool depth of 23 feet with a resultant quencher such mergence of 18 feet. This yields a hydraulic head of 7.8 psi. This incremental pressure yields a local saturation temperature of 234.3^{o}F which would allow operation up to a local temperature of 224.3^{o}F. The HCTL curve therefore has as its 240 psia end point a temperature of 224.3^{o}F. This value of subcooling is known to be quite conservative on two grounds. First the pool tends to develop a stratified vertical temperature profile which would result in a local temperature of several degrees below the bulk pool temperature. The second source of conservatism is the fact that as the pool heats up, water vapor evolves into the wetwell airspace and causes non-condensable flow through the vacuum breakers into the drywell. This process would permit the HCTL criterion to be met at temperatures well above 224.3^{o}F depending on the rate of water vapor evolution and condensation in the containment.

For the high pressure pool temperature which triggers depressurization we have selected 208^{o}F as a conservative limit at a reactor pressure of 1000 psia. Calculations show that the reactor can always be depressurized to remain below the HCTL limit line defined by these two points under any credible reactor condition when the limit is reached. Use of HPCI to maintain inventory and contribute to the depressurization is presumed in this event.

7.2 The Shutdown Transient Model

This calculation consists of eight sub-models which determine the interactions of the various reactor parameters in the shutdown process. These are discussed individually below.

7.2.1 Decay Heat

The reactor transient reduces the reactor power by a factor of three or more in the first few minutes of the transient and the fission power then continues to decrease due to boron injection. The decay heat is carried separately and is based on the initial operating power of the reactor at the transient start and on the assumption of steady operation at that power for a long period of time.

Separation of the decay heat and the fission power is important in that boron addition cannot reduce the decay heat and the decay heat contribution to pool

heating is not negligible. This treatment of decay heat is slightly non-conservative in that it neglects the additional contribution of fissions occurring after the start of the transient to the decay heat level as the transient progresses.

7.2.2 Fission Power

The calculation of fission power in the absence of boron is taken from data presented in NSAC-70 for core power versus downcomer water level, and is used in the analysis for steady state or for slow transients. The validity of this application of the data is supported by comparisons in NSAC-70 of the transient data with steady state calculations. The steady state data points fall very close to the transient data for power versus level.

The power taken from the NSAC-70 data includes both the fission and the decay heat. For this reason decay heat is subtracted from the NSAC-70 data to obtain the fission power. The addition of boron reduces the fission power in inverse proportion to the fraction of hot shutdown boron mixed in the reactor water in the core volume.

This is somewhat conservative since the HSD boron concentration was determined in the absence of voids. This will result in an over estimate of the boron needed for HSD in the presence of decay heat.

7.2.3 Core Flow

For the unborated reactor, data from NSAC-70 may be directly used to determine core flow as a function of level. For the case of boron addition, however, the NSAC-70 data may no longer be used since the boron disturbs the relationship between level and power. For this reason we have chosen to use a flow model which balances the hydraulic head inside the shroud against the corresponding head outside the shroud and equating the difference to irreversible flow losses depending on flow squared.

The NSAC-70 data for core voids and analytically calculated voids in the upper plenum and riser are used to calculate the hydraulic head for a range of flow rates and the NSAC-70 core flow data is used to determine an irreversible loss coefficient as a function of core flow. This model yields exact agreement with the NSAC-70 data for a transient with no boron addition due to the calculation of the loss coefficient from NSAC-70 data.

As boron is added, the change in power permits the model to calculate the change in voids in a consistent manner to describe the reduction in flow. This model is the most critical part of the calculation in terms of the power transient using the current model for boron mixing. This sensitivity results from the calculation of very low flow as boron is added and a corresponding reduction in boron transport rates.

The flow calculation sets a minimum core flow condition such that the core flow may never be less than that required to produce the steam leaving the core. Core inlet enthalpy is considered to be at saturation for this calculation. This assumption is valid as long as the feedwater sparger is uncovered.

7.2.4 Void Models

The voids for the core are determined directly from the NSAC-70 data for average core voids. As boron is added, the portion of the voids due to fission power is reduced in inverse proportion to the fraction of HSD boron in the core.

The unborated void is divided between fission power and decay heat in proportion to their relative magnitude. It is this procedure which introduces conservatism into the estimate for the boron concentration needed for HSD.

Voids in the upper plenum and separator risers are calculated from an analytical model. To assure consistency of the unborated results to the NSAC-70 results, the flow loss coefficient is calculated for the unborated data points to force the model into agreement with NSAC-70. These flow loss coefficients are then used also for calculating the voids when boron is present. For this purpose the loss coefficient is considered to be a function of flow only. Any dependence of the loss coefficients on power cannot readily be derived from the NSAC-70 data and is therefore neglected.

This approach assures maintaining the performance derived by NSAC-70 unchanged, but also permits the influence of boron addition to be approximated.

7.2.5 Steam Flow From the Reactor

The reactor model is based on a constant pressure system, and a calculated steam flow which maintains constant reactor pressure. Reactor internal temperatures remain constant during the transient so that the steam flow is chosen to remove excess heat generation above that required to bring the makeup flow up to saturation.

7.2.6 Makeup Flow

The makeup flow to the reactor is controlled by a rather complex set of logical instructions which consider:

1. The water level
2. The amount and distribution of boron in the reactor
3. The impact on reactor pressure
4. Time

The logic is arranged so that a variety of operator response strategies to ATWS may be simulated. The calculation will not permit a decrease in reactor pressure due to addition of makeup in excess of what the reactor power can bring to saturation. It avoids this by restricting makeup flow to that which can be brought to saturation with no steam flow from the vessel.

7.2.7 Water Level

The water level in the downcomer is calculated from a global mass balance on the vessel. First, the total mass in the vessel is calculated for a new time step. Then the energy change for the time step is calculated. Using the new energy and total mass, the quality is calculated and therefore the mass of

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liquid phase. With the liquid phase mass and the assumption of saturation conditions for the liquid, the core, plenum, and separator voids may be used to calculate the downcomer liquid level.

The assumption of saturated liquid should be valid for those cases where the feedwater sparger is uncovered. Under these conditions the condensation of steam on the cold makeup water plus the mixing with saturated liquid from the steam separators will result in core inlet conditions very near to saturation.

Where the feedwater sparger is covered, this argument is no longer valid. For this condition some degree of subcooling in the downcomer and the lower plenum could occur. In the case of a still critical system this could result in increased fission power and in the case of a shutdown reactor it could cause the calculated steaming rate to be high for some period of time. We believe that the assumption of saturation does not have a major influence on the results of the calculations performed.

7.2.8 Boron Mixing

Our initial intent in the conduct of our calculations was to use the results of the General Electric evaluation of their 3D mixing test data for standpipe injection. This data is reported in NEDE-22267 and the data analysis and correlation is reported in NEDC-30921.

In the latter report GE did not develop a model from the data, but rather developed a functional fit to the data at each of four constant flow rates which yield a "mixing coefficient" as a function of time. This function, however, applies only to constant flow situations and cannot be used in a transient where flow varies. For this reason, it was necessary to develop a new model for boron mixing which could be adjusted to fit the data.

A boron mixing analytical model was developed which was intended to yield agreement with the GE evaluated data when proper values of the model parameters were chosen. The model divides the reactor liquid phase volume into four regions:

1. The upper portion of the lower plenum into which the boron is injected.

2. The core and bypass volume.

3. The upper plenum, separator, downcomer, and jet pump volume.

4. The lower portion of the lower plenum in which stagnated boron may accumulate.

In addition each of these subvolumes could be further divided into sub-nodes (up to 10) except for the fourth volume above.

The model used data for entrainment efficiency and a remixing time constant for the "stagnant" volume developed by General Electric for the BWR Owner's Group and reported in NEDC-22166 (August 1983). In addition, a core bypass model was added to represent the GE approach to data interpretation.

The basis for the core bypass model was that GE excluded the outer-most ring of instrumented fuel bundle mockups from the data reduction and also the inner ring bundle nearest the injection stand pipe. Since some of these channels may see greater than average boron concentrations on the first pass of fluid as it circulates around the reactor flow path, some boron effectively bypasses the core on its first circuit of the reactor volume. The bypassed boron should mix relatively uniformly beyond the core, and so will be seen in subsequent fluid passes. This was modeled by bypassing a portion of the entrained boron around the core and injecting it at the core exit.

The model then provides for boron injection at three locations. The upper lower plenum, the upper plenum, and the lower lower plenum. The bypass coefficient, the mixing efficiency, and the remixing coefficient (which removes stagnated boron) were all modeled as functions of flow. Starting with the data for these functions from the original model a variety of bypass functions were tried without success. After extensive attempts to obtain agreement by drastic adjustments to the three functional relationships two extreme cases were calculated: first, zero mixing efficiency with adjustments to the remixing time constant to yield best agreement, and, second, one hundred percent mixing efficiency with one hundred percent bypass. For the first of these, the bypass model has no influence on the results and for the second the remixing time constant has no practical influence on the results.

The finding, for these two extreme cases, was that both produced nearly identical results when the functional forms giving the best fit to the data were selected.

For the first case, a remixing time constant much slower than that recommended by General Electric was required to yield the best agreement for the 5% flow test. For higher flow rates, the long term values for mixing coefficient calculated were reasonable, but the short term values were much too high compared to the evaluated data.

For the second case, ten subnodes each in the downcomer, the upper lower plenum, and the core were used to represent the circulation delay. Finer noding did not further improve the results. Much coarser noding could be used in the core and upper lower plenum. The best fit obtained by this approach had exactly the same characteristics as the first approach. The 5% curve fit was conservative. The higher flow results were good in the asymptotic region, but much higher than the General Electric correlation in the "knee" region of the curve.

From these results, it was concluded that no physically reasonable combination of functional dependences for the three mixing phenomena incorporated into the model could yield an improved fit to the correlation.

At this point we investigated the mixing behavior of fluid jets into relatively stagnant fluid volumes. Data on entrainment of surrounding fluid by jet streams indicates that a very high degree of mixing of the boron solution with the reactor water is to be expected as it leaves the standpipe. The exit velocity at 86 gpm is about 70 feet/second through 8 holes of 0.25 inch diameter. Perry (Table 5-6) indicates that the mixing out to 100 diameters (25 inches) for either injection rate, 43 gpm or 86 gpm, should mix the boron solution with entrained reactor water to bring the mixture density to within a few percent of reactor water density.

[illegible] from [illegible]

There are reservations about this conclusion due to qualifications expressed in the discussion in Perry. The fact that a liquid (as opposed to test results for air) is involved and that the jet is a denser fluid could both reduce the mixing efficiency. Nevertheless, we believe that this result supports the use of a mixing efficiency near 100% independent of the reactor flow rate. Only at the very lowest flow rates would any discernable degree of stagnation be expected due to the thorough mixing due to jet entrainment. This apparent stagnation would be primarily due to the low bulk fluid velocity.

Examination of the lower plenum thermocouple data from the 3D tests support this conclusion. All lower plenum thermocouples show a start of the temperature increase that is nearly the same for all and a rate and magnitude of temperature rise that is nearly the same for all. This behavior appears to hold for all tests from 0% flow to 20% flow. The only conclusion that can be drawn from this behavior is that mixing in the lower plenum is rapid and reaches nearly the entire lower plenum volume.

This finding supports the GE conclusion that mixing is independent of injection rate (as the theory states it should be), but does not support the GE concept of stagnation of the injected boron except possibly for the case of zero core flow. Even in the zero flow case, however, the evidence clearly demonstrates that the lower plenum is well mixed, but perhaps not perfectly mixed.

On the basis of the above discussion we have chosen to utilize a 100% mixing efficiency in our model combined with a 100% bypass of the core on the first circuit. The bypass is intended to represent the data censoring concept used by GE in interpreting the data. We believe that the following conservatisms result.

1. The time of boron entry into the core is delayed by from 75 seconds at 20% flow to 300 seconds at 5% flow using a bypass fraction of unity.

2. The entire downcomer volume has been assumed to be included for boron mixing when it is believed that little mixing with the volume below the jet pump throat will occur.

3. Instantaneous and uniform mixing within each sub-node is assumed.

These conservatisms are believed to have a strong influence on the results of the analysis and are believed to more than compensate for potential non-conservatisms in the model. The non-conservatisms are:

1. Complete mixing of the injected boron is assumed with no stratification of dense solution.

2. Concentration within the core is assumed uniform.

3. The recirc loop volume is excluded from the calculation.

The first of these is believed to be fully supported by the nature of the 3D test data and by theoretical and test results for jet entrainment. The second

is believed unimportant in that the injected boron is not counted as in-core boron on its first pass, and concentration should be nearly uniform after passage through the upper plenum, separator, and jet pumps. Exclusion of the recirc loop volume is justified on the basis that no circulation through the loop is expected during the boron injection period. Flow induced by the natural circulation flow is expected to be small and the inclusion of the lower downcomer volume should compensate for any flow induced.

In order to establish an upper bound to SLCS performance a second set of calculations was performed in which a mixing efficiency of 100% was assumed, but no core bypass was assumed. Since all boron injected into the lower plenum must now pass through the core before reaching the downcomer volume, mixing coefficients much greater than unity are calculated in the first minute or so of the injection transient, and the mixing coefficient approaches unity from above asymptotically. The time delay for boron circulation through the reactor is properly represented in this model, however. The apparent large improvement due to high values of mixing coefficients does not really have a strong impact on the overall pool temperature transient since the boron concentrations early in the transient are small. The major benefit from this model is a result of the asymptotic behavior which results in core boron concentrations 10% to 30% higher at longer times in the transient.

The two sets of calculations performed represent upper and lower bounds on the reactor response to an ATWS transient. Based on our review of the General Electric test data in NEDE-22267, we believe that the zero bypass case is the better model for actual system response.

8.0 Results of the Analysis

A summary of the results of the calculations performed is shown in Table 8-1. The calculations have considered both the existing 43 gpm boron injection system and the future 86 gpm system. The limiting cases for core bypass, 0 and 1, are believed to represent bounding cases for reactor response to a limiting isolation ATWS. Four response strategies have been evaluated. These are:

1. BWROG response. Terminate make up flow until TAF, then maintain level at TAF.

2. Same as 1 above except control level at -110 inches to avoid level 1 trips and loss of wide range level indication.

3. Allow full flow HPCI/RCIC operation until level begins to increase due to boron injection, then maintain that level.

4. Allow full flow HPCI/RCIC operation until water level returns to normal. Maintain level at normal.

An evaluation of these results follows.

<u>Boron Injection at 43 gpm</u>

Initiation of boron injection is assumed to begin at 120 sec. Since the SLCS transit time for boron is only 13 seconds, it has been neglected both for the

[illegible]

[illegible]

[illegible]

1. [illegible]

2. [illegible]

3. [illegible]

4. [illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

43 gpm and 86 gpm cases. All four response strategies avoid reaching the ATWS HCTL curve before shutdown, and, on that basis, all four response strategies are acceptable. This is an important result in that if the operator fails to throttle HPCI for strategy 3, the resulting pool temperature does not exceed HCTL. Table 8-2 shows that for strategies 1 and 2 there are four actions required within 160 seconds while for 3 only two are required. The second of these actions for strategy 3 is not critical in that failure to execute it will still yield acceptable results.

The TAF case is not considered acceptable on the basis that level 1 isolations and initiations would be generated prior to 160 seconds and would as a result cause a much heavier burden on the operators as a result of the L1 initiations. Those initiations can be avoided by holding level at a minimum value of -110 inches instead of TAF.

The resulting advantage of strategy 2 over strategy 3 is between 13°F and 17°F. This advantage is a result of the fact that mass of water is less for 2 than for 3. In both cases HSD is reached within 30 seconds after the peak suppression pool temperature is reached. As a result, the remixing guidance of the EPGs does not influence the transient, since HSD is reached before make up flow can actually be increased.

Even for the case of a 43 gpm SLCS the concerns over reducing water level beyond the HPCI/RCIC equilibrium level are considered to outweigh the slight additional reduction in peak pool temperature that results. Furthermore, an advantage of this magnitude is strictly temporary since the SLCS injection capacity will be increased to 86 gpm in the future.

Boron Injection at 86 gpm

Increasing the boron injection rate to 86 gpm essentially removes any concern over exceeding the HCTL curve regardless of the response strategy. The advantage of strategy 2 over strategy 3 is reduced to the range of 5°F to 8°F. Performance is so improved that the failure to throttle HPCI flow for strategy 3 will not result in exceeding even the 165°F HCTL value for non-ATWS events. For the higher injection rate the incentive to operate the reactor at low water levels to reduce suppression pool peak temperature has essentially been removed. The incentives to avoid operation at low water level remains.

Injection Rate	Bypass	Strategy*	Maximum Pool Temperature	Time To: HSD	Latest SLCS Start	L2	110°F	145°F	165°F	Minimum Level
43	0	1	153.5	1000	1075	95	90	560	--	160
43	0	2	163.4	1020	660	95	90	410	--	150
43	0	3	180.0	1190	420	95	90	370	650	170
43	0	4	196.8	1510	240	95	90	370	650	--
43	1	1	165.2	1320	860	95	90	450	1220	160
43	1	2	174.3	1370	530	95	90	390	800	150
43	1	3	187.1	1570	340	95	90	360	620	190
43	1	4	201.6	1590	170	95	90	360	620	--
86	0	1	137.0	510	1365	95	90	--	--	160
86	0	2	141.7	520	910	95	90	--	--	140
86	0	3	149.9	640	740	95	90	430	--	200
86	0	4	157.1	790	710	95	90	450	--	--
86	1	1	148.6	800	1190	95	90	550	--	160
86	1	2	152.1	850	800	95	90	460	--	150
86	1	3	157.3	900	680	95	90	390	--	220
86	1	4	161.6	870	655	95	90	400	--	--

* 1 = Hold level at TAF.
2 = Hold level at -110 inches.
3 = Full HPCI/RCIC flow to equilibrium level, then hold.
4 = Full HPCI/RCIC flow to NWL.

Table 8-1

Summary of Calculated Results
for Response Strategies

	Strategy			
	1	2	3	4
1. Operator trips HPCI/RCIC	90	90	--	--
2. Operator initiates SLCS	120	120	120	120
3. Operator restarts HPCI	160	150	--	--
4. Operator adjusts HPCI flow	160	150	170	--
5. Operator starts RHR	450	450	450	450
6. Operator closes HX bypass	1160	1010	970	970

Table 8-2

Timing Requirements for Operator Actions (seconds)

(43 gpm Boron Injection)

EVALUATION OF SSES DEVIATIONS

FROM BWR OWNERS GROUP EPG'S

(NON - ATWS)

Prepared By: ______________________
Sr. Proj. Engr. Nuclear Systems

Reviewed By: ______________________
Supervisor - Eng. Aval.

Approved By: ______________________
Manager NF&SE

Scope

The scope of this document is to provide an evaluation of the differences between the BWR Owner's Group Emergency Procedures Guidelines (EPG's) and the Susquehanna Steam Electric Station (SSES) specific EPG's which have been generated. This review is limited to procedure differences which do not relate to the anticipated transient without scram (ATWS) event. The following differences are evaluated in this report.

1. Water level at which the Reactor Pressure Vessel (RPV) control sequence is entered.

2. Allowing a main steam isolation valve (MSIV) to reopen during a non-ATWS event.

3. Additional guidance on Site Area Emergency dose rates as entry into RPV control sequence.

4. A deleted step in secondary containment control which gives guidance for secondary containment high water levels.

5. Allow the operator to use the bypass valves, if available, in emergency depressurization along with the safety/relief valves. (S/RV's).

Background & Purpose

The NRC has issued a Safety Evaluation Report (SER) on the EPG's Revision 3 as submitted by the BWR Owners Group, of which PP&L is a participating member. However, in formulating the SSES specific EPG's, design differences and plant specific features at SSES required some changes in the Owners Group generic EPG's to more accurately reflect the SSES design. A requirement in the SER written by the NRC allows the Generic EPG's to be modified if an evaluation of the consequences of such changes is included. The purpose of this document is to provide an evaluation of the non ATWS related differences denoted above.

Analysis

Change Entry Level into RPV Control Guideline from L3 toL2

The level for entry into the RPV Control Guidance procedures as given in the generic EPG's is the level at which a scram on low water level occurs (L3). A review of plant startup data (NPE Technical Report NPE-85-001) shows that under a turbine trip condition from normal water level, void collapse in the core will result in the narrow range water level reaching L3. Since the EPG's should be entered into in emergency conditions only, an entry level on L3 could refer the operator to the EPG's under normal conditions. Therefore, a better entry level into this guidance would be the level at which high pressure make up systems are actuated or L2. A transient in which the level fell such that the high pressure injection systems are initiated would be indicative of level control problems in normal plant systems, therefore, setting the entry conditions at L2 for the RPV Control guidelines is technically justified.

Allow the Use of the Main Condenser Under Non-ATWS Events

The generic EPG's allow the operator to attempt to reestablish steam flow to the condenser under ATWS conditions when the condenser is available and no indication of gross fuel failure exists by opening the main steam isolation valves (MSIV's). At SSES, reestablishment of the main condenser, if available, as a source of cooling is the desired cooling mode under ATWS or non-ATWS conditions provided that no indication of fuel failure exist. Since fuel failure is the only condition under which long-lived isotopic fission product contamination can migrate to the condenser, the requirement that no indication of fuel damage exist prior to reestablishment of condenser cooling adequately precludes the possibility of radiation due to fission product transport reaching the condenser and therefore transmitted to the environment. Under conditions which indications of fuel damage exist, the MSIV's remain closed and cooling is established via the suppression pool cooling path. If fuel damage does occur or is undetected, the jumpering open of the MSIV's does not preclude the closing of the valves on high main steam line radiation levels, according to operations staff.

Therefore, it is technically justified to delete the condition that boron injection be required before the MSIV's are to be reopened. Deleting this requirement allows reestablishment of condenser cooling under appropriate conditions under both ATWS and non-ATWS conditions in procedural step RC/P-1.

Add an Entry to RPV Control Guideline on the Basis of Site Emergency Declaration

The generic EPG's have as entry level into the RPV control guideline the criteria for a general emergency as far as boundary dose calculations are concerned. At SSES, this guidance is transformed into a slightly more conservative approach, still within the intent of the EPG's, whereby the entry condition into the RPV Control Guideline is off-site dose equivalent to a site emergency rather than a general emergency. By taking control of reactor pressure and reducing it, the general emergency classification may be avoided by reducing the driving force for fission product transport. Since the RPV control guideline at SSES would be entered before the generic EPG's would have the operator enter the guideline, the SSES EPG's are consistent with the intent of the generic EPG's and are applied conservatively. Thus, no safety impact due to this addition exists.

Delete Section in Secondary Containment Control

The generic EPG's contain a section, SC/L. which instructs the operator to perform actions based on the level of water existing on the floor of rooms in secondary containment. At SSES, no water level measurements are available for rooms in secondary containment prior to the alarm condition. However, the location of safety related equipment in secondary containment is such that their function is maintained even at the alarm setpoint. Therefore, operator response to the alarm is sufficient to maintain function or at least maintain control. The action steps required to respond to alarm conditions are covered in SSES off-normal procedures, therefore no specific steps from the EPG's are required.

Use of Bypass Valves During Emergency Depressurization

Under Emergency Depressurization conditions, the generic EPG's instruct the operator to open six safety/relief valves (S/RV's) (either all of the ADS valves or enough non-ADS valves to equal a total of six valves open). The SSES specific EPG's provide the same general instructions but allow the use of the bypass valves if prerequisites are met.

The prerequisites for maintaining the condenser available are the same as the requisites given for opening the MSIV's as discussed earlier in this section. The main concern is to limit the amount of fission product transport out of containment into the condenser. If no indication of fuel damage exist as a prerequisite, the condenser may be used as a heat sink and thus may be used during depressurization. If fuel damage indications are present, then only the S/RV's opening to the suppression pool should be used as a depressurization path.

The calculation for equivalencing S/RV's to bypass valves has been reviewed and found to be acceptable.

Conclusion

The differences between the generic EPG's and the SSES specific EPG's have been reviewed. In each case the SSES specific EPG's were found to be acceptable from a safety standpoint.

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EVALUATION OF SSES DEVIATIONS

FROM BWR OWNERS GROUP EPG'S

(NON - ATWS)

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